Fast Reactor Fuels

Steven L. Hayes and Douglas L. Porter

Nuclear Fuels and Materials Division
Fuel Performance and Design Department

June 5, 2009

Idaho National Laboratory

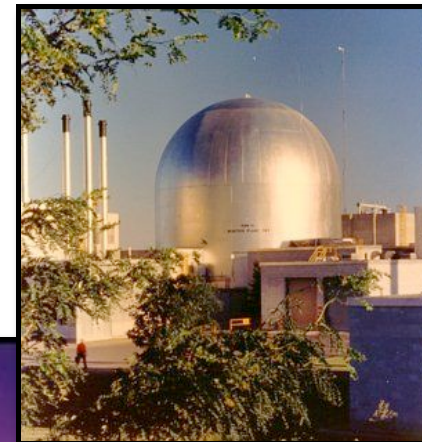
Outline of Presentation

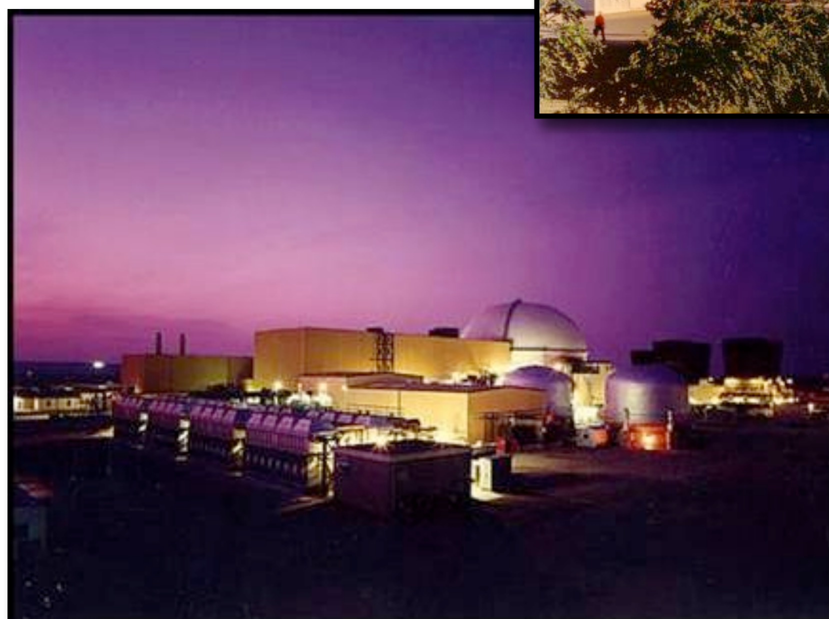
- **Introduction**
- **SFR Fuels Experience in the US**
  - Fuel Types
  - Fuel Performance Issues
  - Experience/Testing
- **Experience with Fuels Containing Minor Actinides**
- **Summary**

SFR Fuels Experience in the US

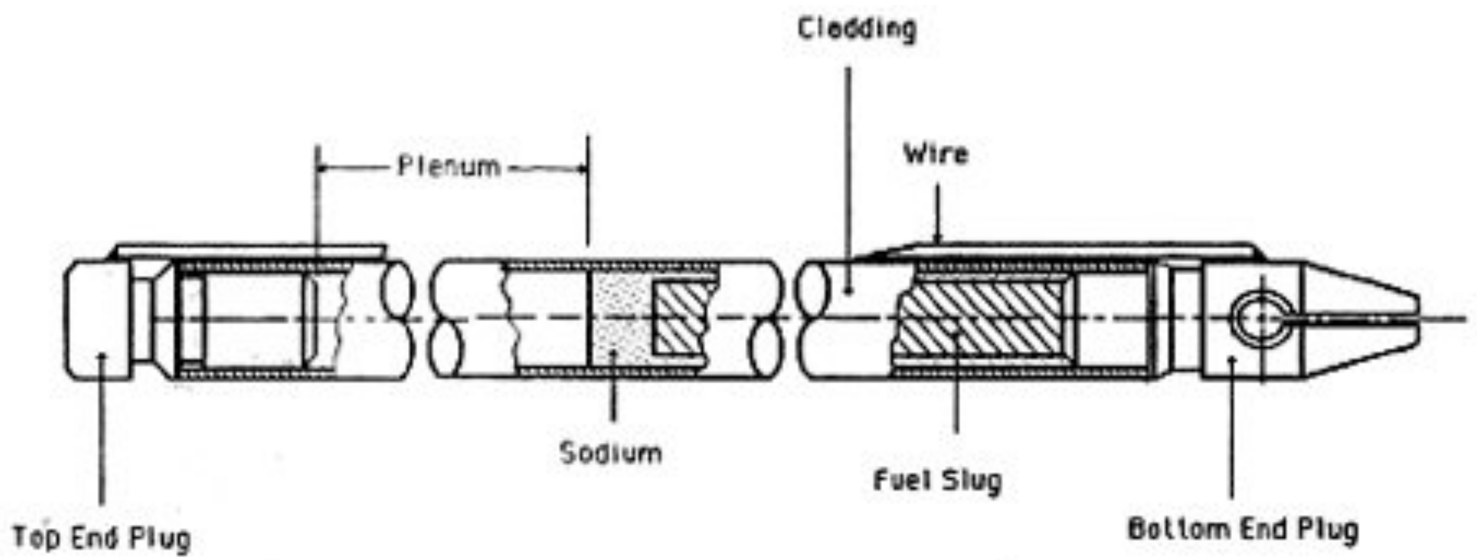
SFR Fuels Experience in the US

- **Metallic Fuels**
  - EBR-I, Fermi-1, EBR-II, FFTF
  - U-Fs, U-Mo, U-Zr, U-Pu-Fs U-Pu-Zr, others
- **Mixed Oxide Fuels (MOX)**
  - EBR-II, FFTF
  - $(U,Pu_{0.2\text{-}0.3})O_2$
- **Mixed Carbide Fuels (MC)**
  - EBR-II, FFTF
  - $(U,Pu)C$ w/15% $(U,Pu)_2C_3$



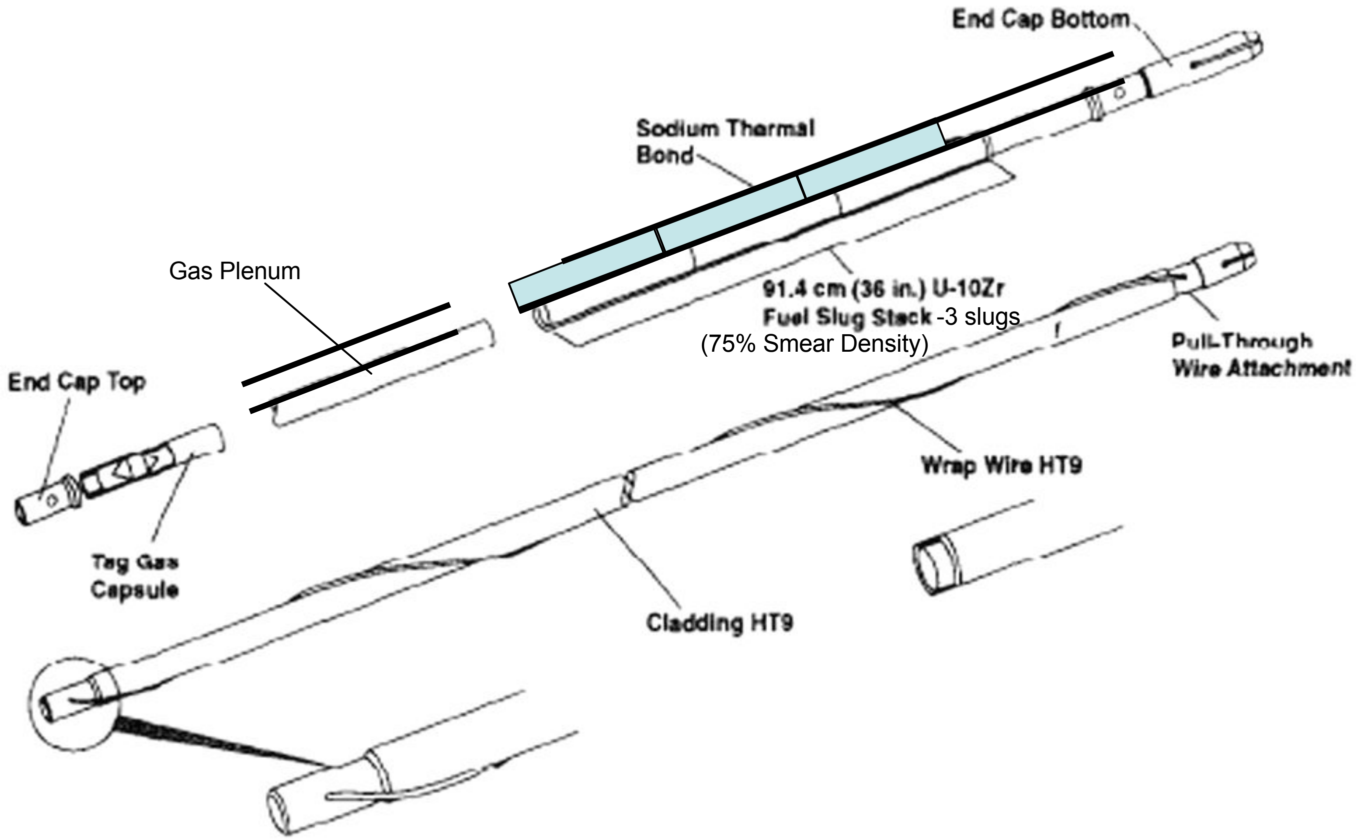


Metallic Fuel Design (EBR-II)



Features of a Metallic Fuel Pin (from Pahl, et al, 1990)

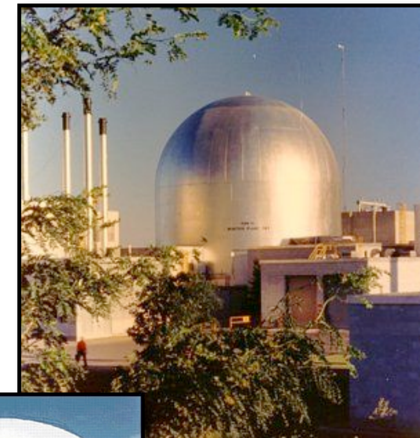
Metallic Fuel Design (FFTF)

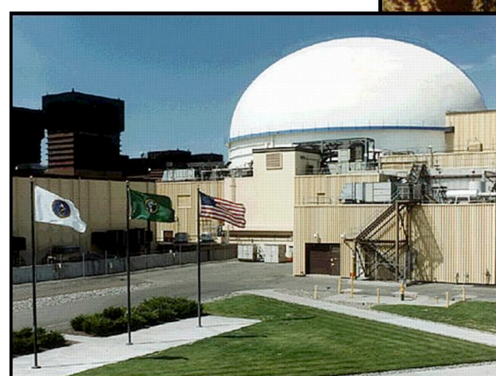


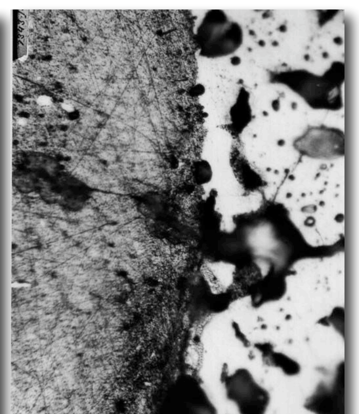
FFTF Series III.b Metallic Driver Fuel Design (from Pitner and Baker, 1993)

Important Metallic Fuel Performance Phenomena

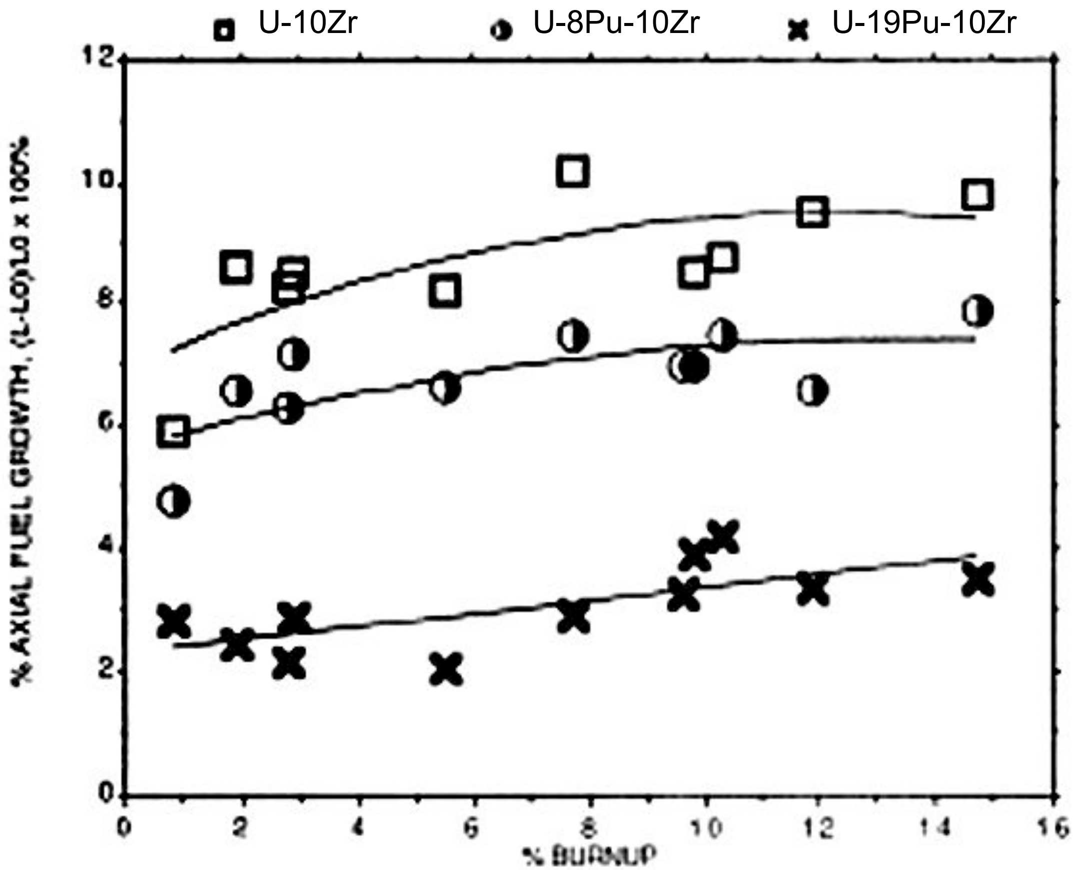
- Irradiation growth
- Fuel swelling and fuel-cladding mechanical interaction (FCMI)
- Gas release
- Fuel constituent redistribution
- Fuel-cladding chemical interaction (FCCI)





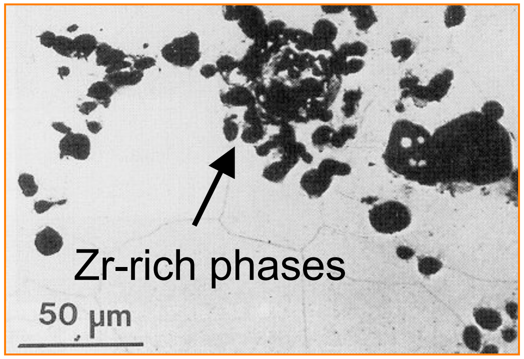


Metallic Fuel Behavior—Axial Growth

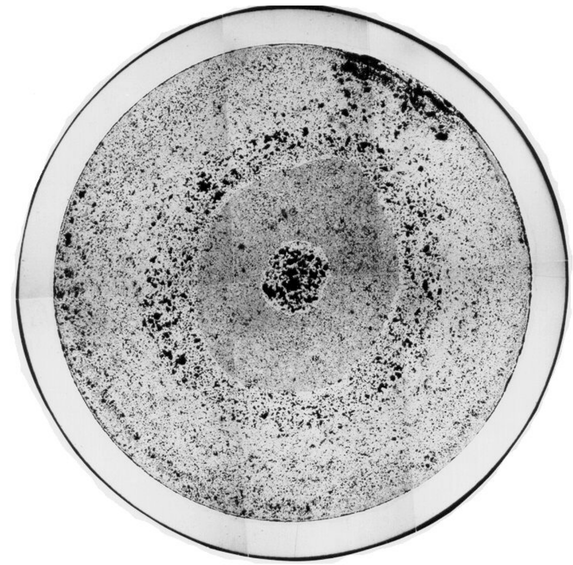


Axial Fuel Growth, from Pahl et al, 1990

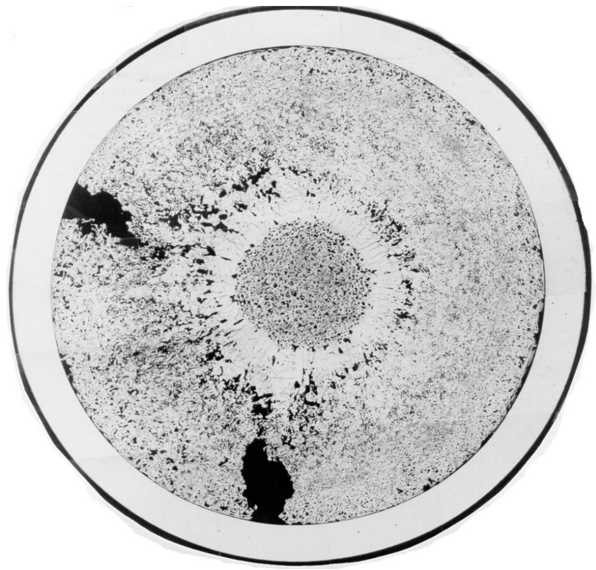
Metallic Fuel Behavior—Swelling & Restructuring



As fabricated U-20Pu-10Zr



X420B at 17% BU



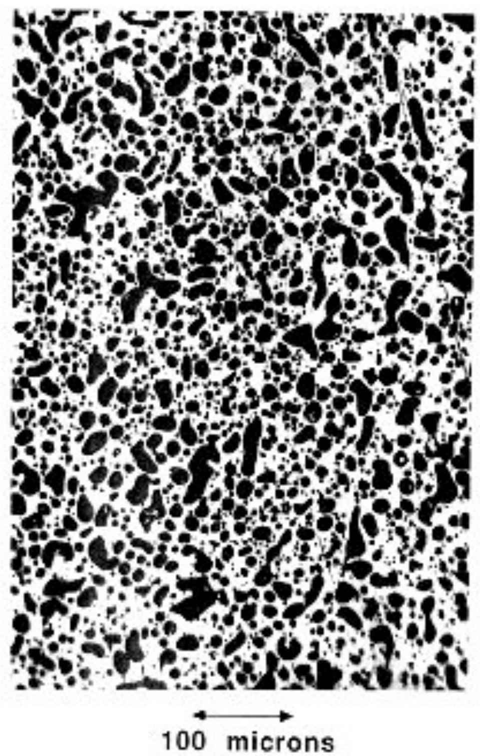
X419 at 3% BU

X423A at 0.9% BU

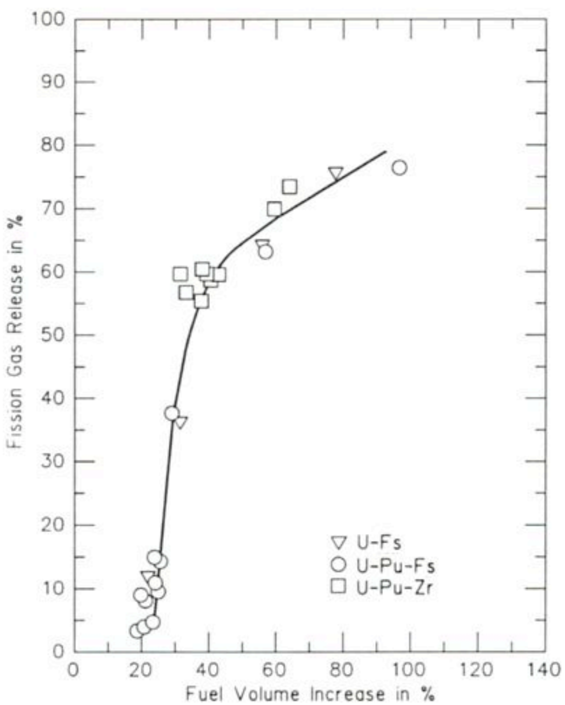
- Redistribution of U and Zr occurs early
- Inhomogeneity doesn't affect fuel life

Metallic Fuel Behavior—Swelling & Gas Release

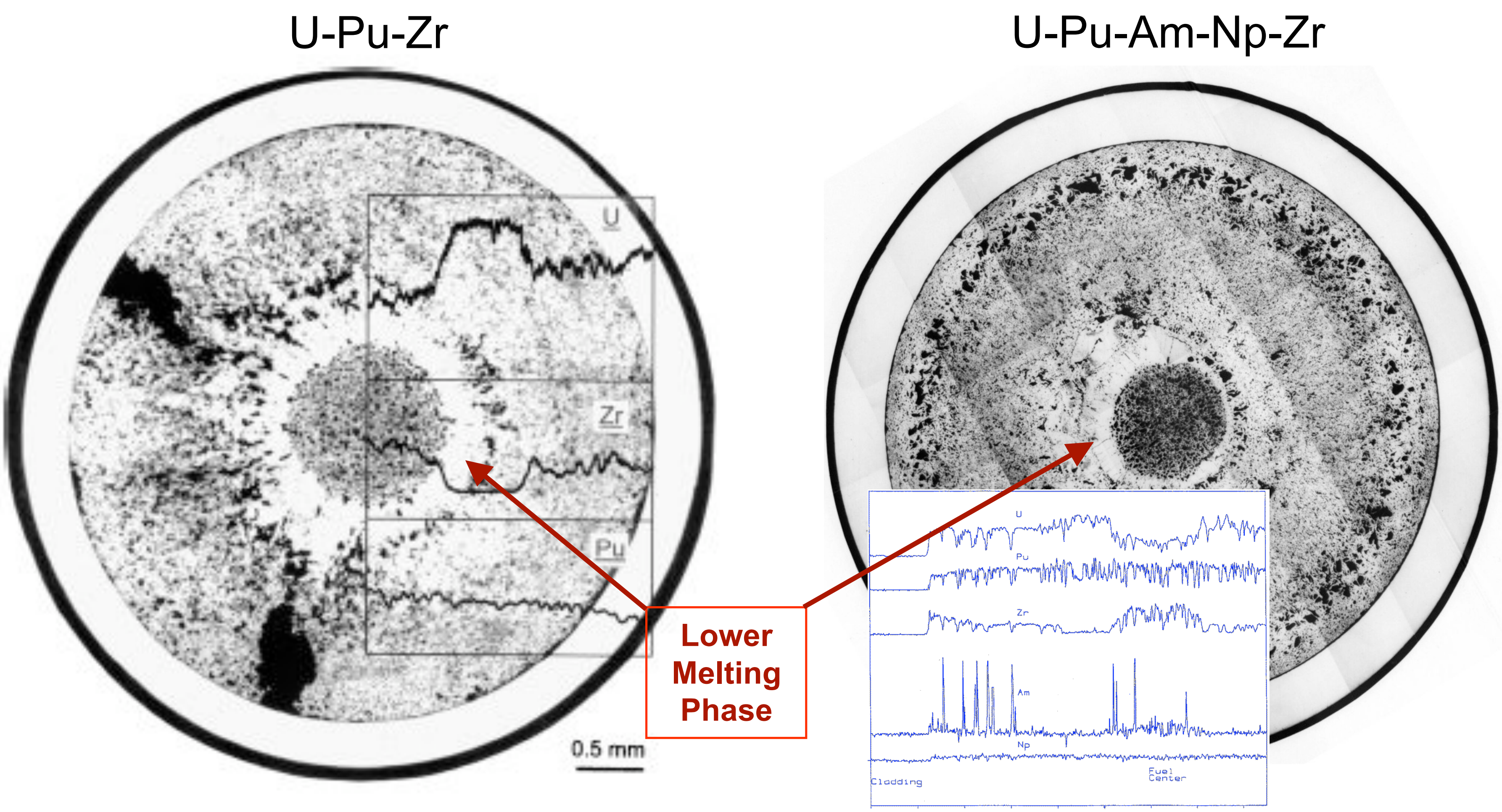
- **Swelling**
  - Low smear density fuels
  - Rapid swelling to 33 vol% at ~2 at.% burnup
- **Gas Release**
  - Inter-linkage of porosity at 33 vol% swelling results in large gas release fraction
  - Decreases driving force for continued swelling



U-19Pu-10Zr (γ-phase) at 2 at.% burnup



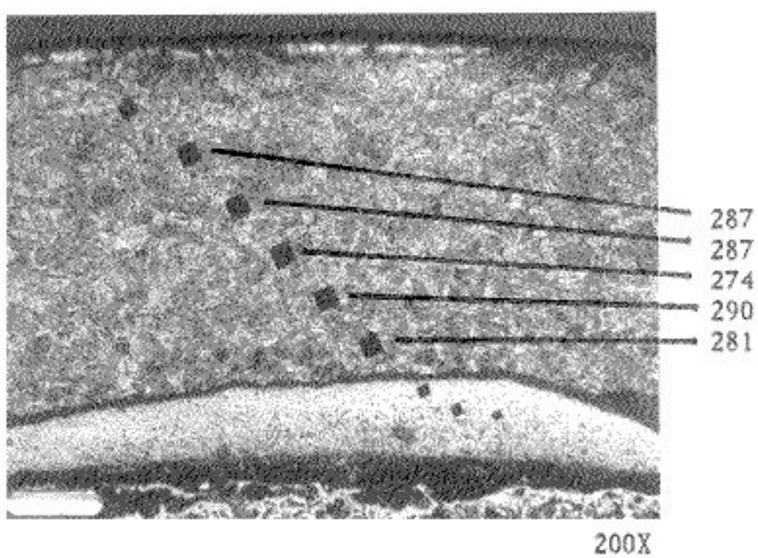
Metallic Fuel Behavior—Fuel Constituent Redistribution

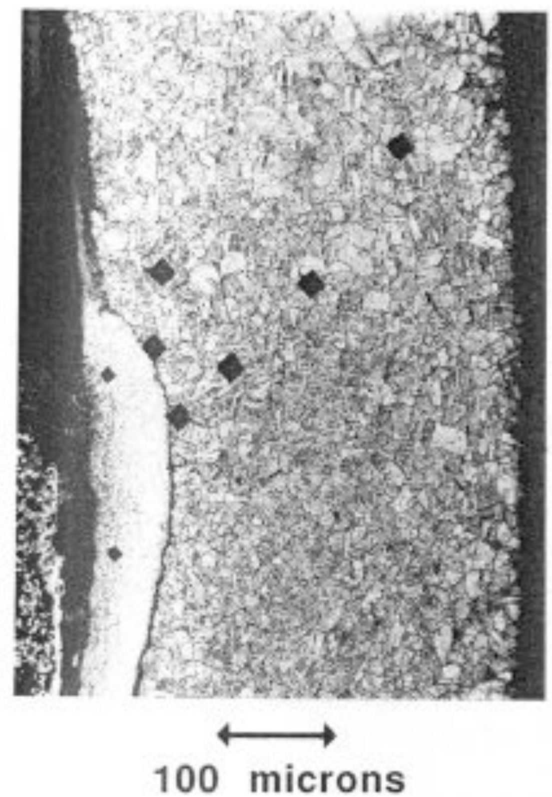


Metallic Fuel Behavior—Steady-state FCCI

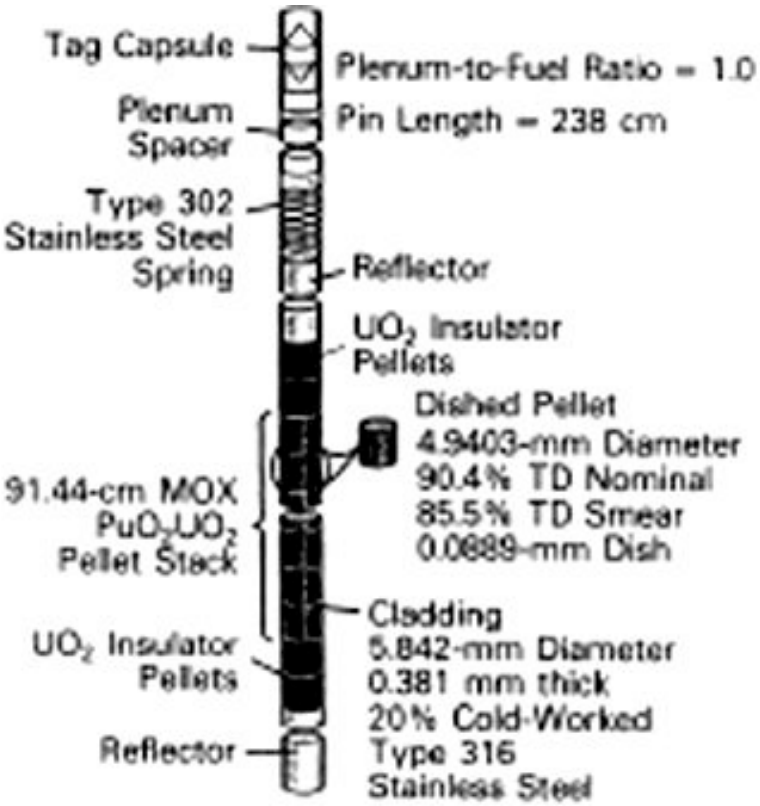
- **Fuel-Cladding Inter-diffusion**
  - RE fission products (La, Ce, Pr, Nd) and some Pu reacts with SS cladding
  - Interaction product brittle
  - Considered as cladding wastage

U-19Pu-10Zr with D9; 12 at.% burnup (from Pahl, et al, 1990)





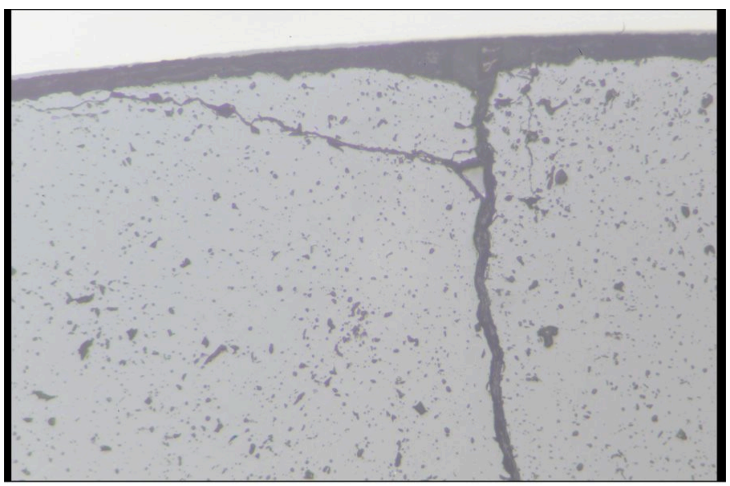
MOX Fuel Design (FFTF)



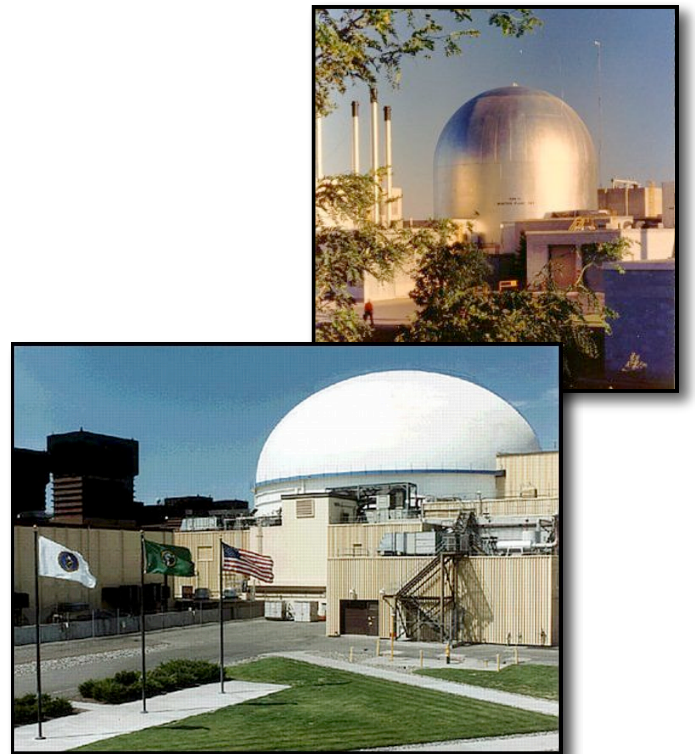
FFTF He-bonded MOX Fuel: a) Driver Fuel and b) Core Demonstration Experiment Fuel
(from Bridges et al, 1993)

Important MOX Fuel Performance Phenomena

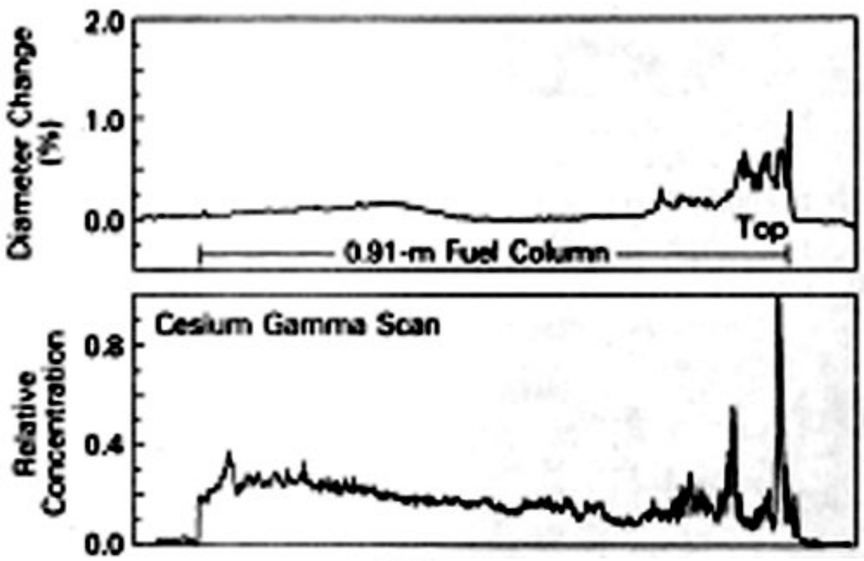
- Fuel swelling and FCMI
- Fuel restructuring
- Gas release
- FCCI
- Fuel-coolant compatibility



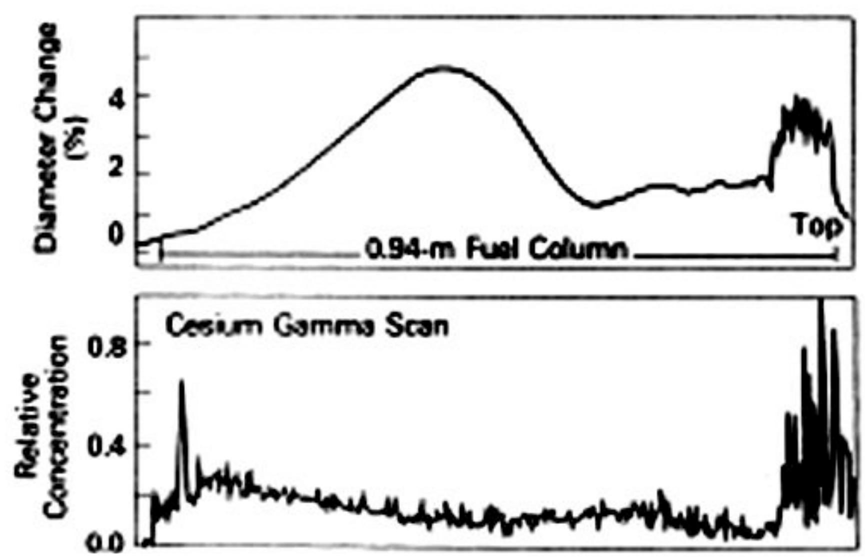
PROF-3 G 50 um



MOX Fuel Behavior—Fuel Swelling and FCMI



(a)

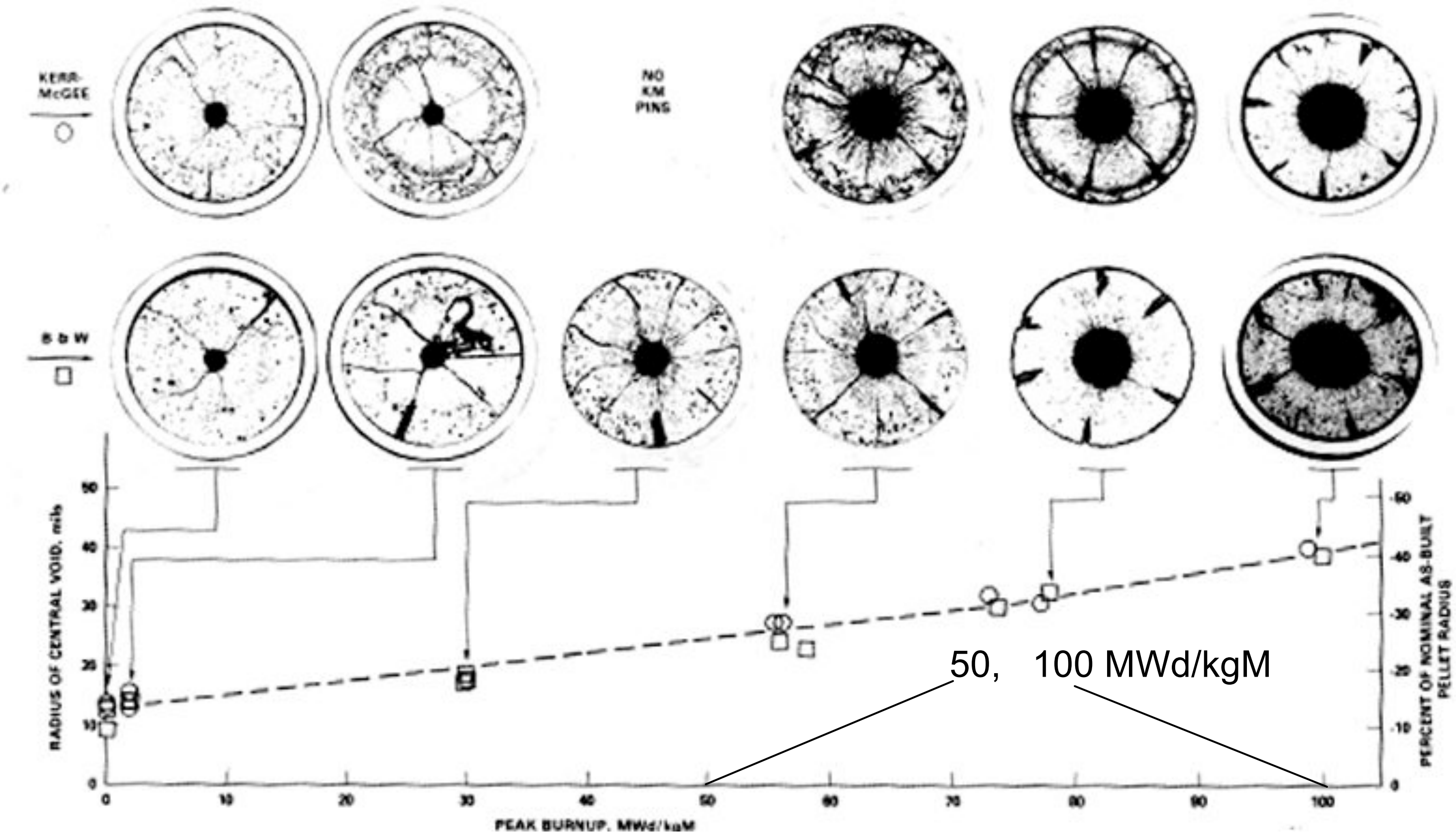


(b)

Diameter and cesium fission product accumulation in high-temperature MOX pins, HT9-clad (a) and D9-clad (b). Cs interacted with MOX fuel causing FCMI.

(from Bridges, et al ,1993)

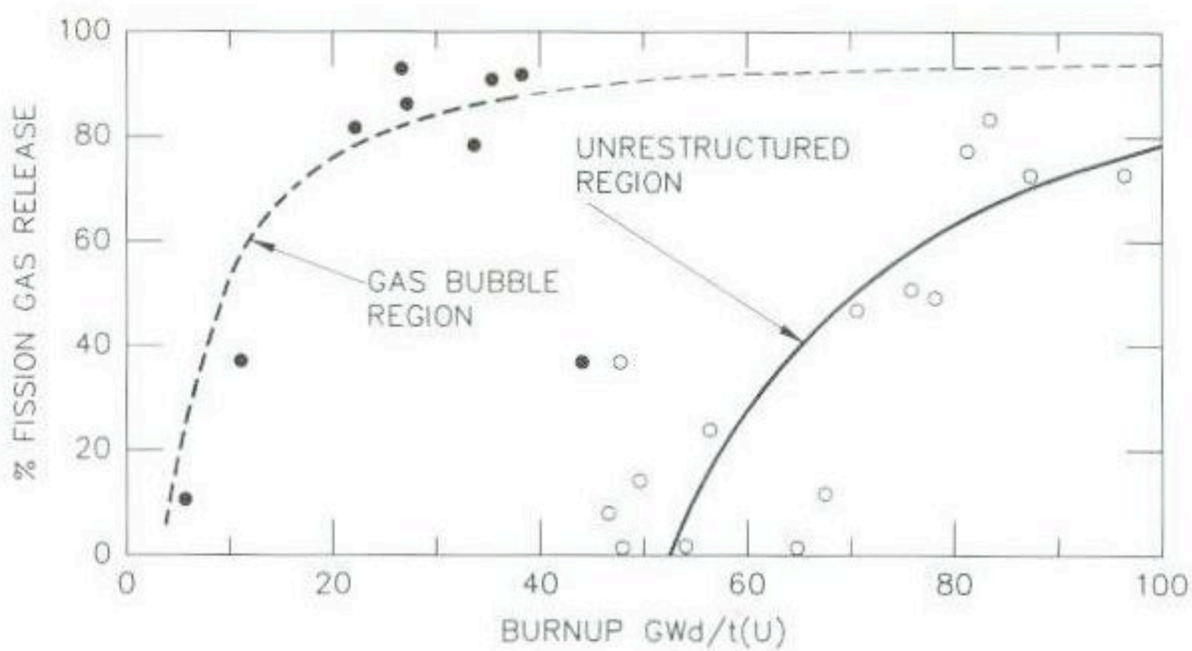
MOX Fuel Behavior—Restructuring



MOX fuel ceramography of FFTF driver fuel produced by Kerr-McGee and Babcock and Wilcox, showing restructuring as a function of burnup. (from Hales, et al, 1986)

MOX Fuel Behavior—Gas Release

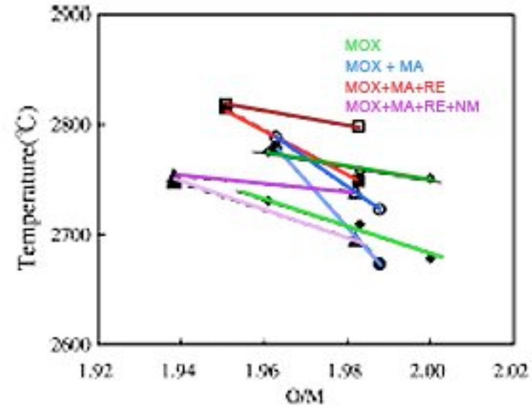
- **MOX fuel operated at high temperature and undergoing restructuring exhibits high gas release.**



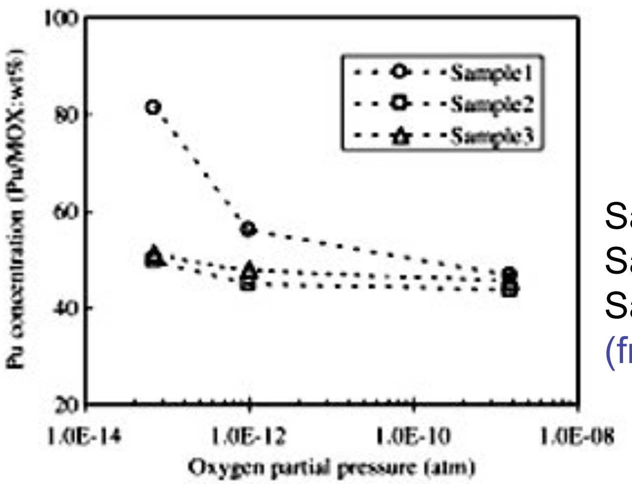
(from Lambert, et al, 1994)

MOX Fuel Behavior—FCCI

- **Hypostoichiometric MOX for SFRs**
  - As-fabricated O/M < 2.00 to suppress free oxygen at high burnup, mitigate FCCI
  - O/M ratio affects fabrication
  - O/M ratio affects properties



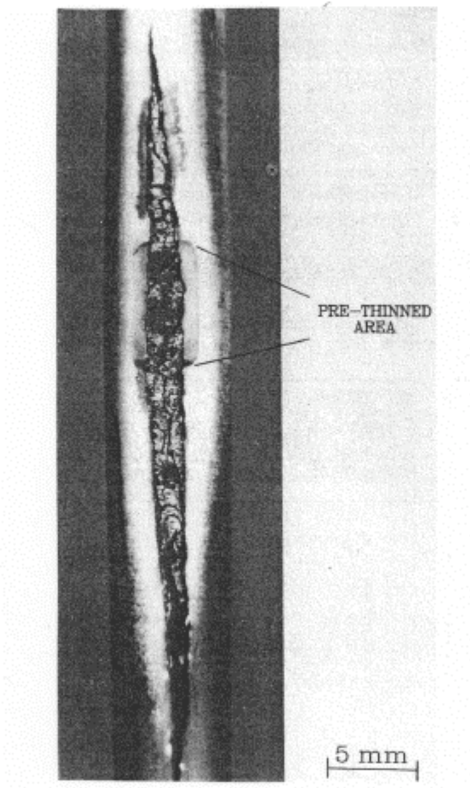
Melting T vs O/M
(from Morimoto, et al, 2005)



Sample 1 – MOX + MAs
Sample 2 – MOX+MAs+REs
Sample 3 – MOX+MAs+REs+NMs
(from Morimoto, et al, 2005)

MOX Fuel Behavior—Fuel-coolant Compatibility

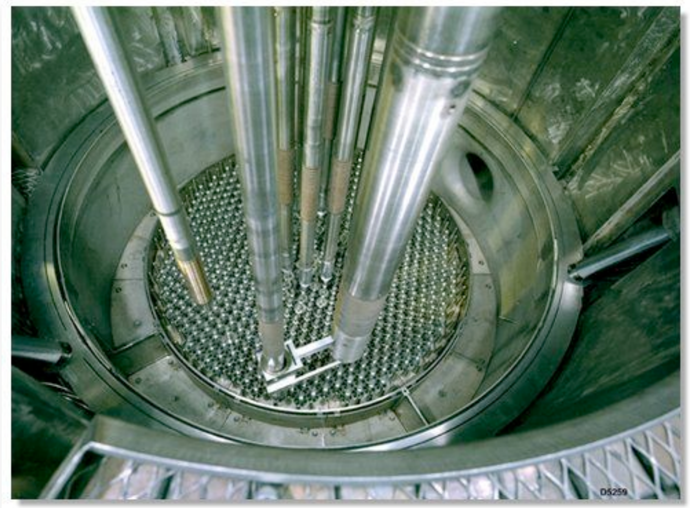
- Run-beyond-cladding-breach (RBCB) of MOX accompanied by fuel/Na reaction and initial crack extension
- Fuel loss can be related to degree of interaction.
- Reactant layer becomes coherent and inhibits further reaction with coolant.

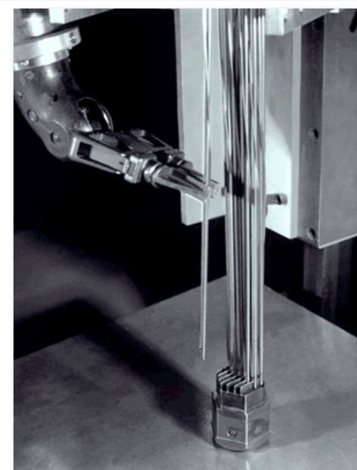


Typical breach extension in induced midlife failure, EBR-II K2B test.
(from Lambert, et al, 1990)

Stainless-Steel Cladding & Duct Performance

- **Performance Issues**
  - Cladding dilation
  - Duct dilation, bowing, or twisting
- **Irradiation Behavior**
  - Void swelling (AS)
  - Irradiation creep (AS & FMS)
  - Irradiation embrittlement (AS & FMS)
- **Alloys to Address Issues**
  - Advanced austenitic stainless steels
  - Ferritic & tempered-martensitic stainless steels
  - Oxide-dispersion strengthened steel alloys





Base Fuel Technology: US Experience

Crawford, Porter, Hayes, **Journal of Nuclear Materials**, **371**: 202-231 (2007).

	Metallic	Mixed Oxide	Mixed Carbide
Driver Fuel Operation	≥ 120,000 U-Fs rods in 304LSS/316SS 1-8 at.% bu ~13,000 U-Zr rods in 316SS 10 at.% bu	>48,000 MOX rods in 316SS (Series I&II) 8 at.% bu	None applicable
Through Qualification	U-Zr in 316SS, D9, HT9 ≥ 10at.% bu in EBR-II & FFTF	MOX in HT9 to 15-20 at.% bu (CDE) MOX in 316SS to 10 at.% bu	None applicable
Burnup Capability & Experiments	600 U-Pu-Zr rods; D9 & HT9 to > 10 - 19 at.% in EBR-II & FFTF	4300 MOX rods in 316SS to 10 at.%; fab var's; CL melt 3000 MOX rods in EBR-II; peak at 17.5at.% bu 2377 MOX rods in D9 to 10-12 at.% bu; some at 19 at.% bu	18 EBR-II tests with 472 rods in 316SS cladding; 10 rods up to 20 at.% w/o breach 5 of which experienced 15% TOP at 12 at.% 219 rods in FFTF, incl 91 in D9, 91 with pellet & sphere-pac fuel
Safety & Operability	6 RBCB tests U-Fs & U-Pu-Zr/U-Zr(5) 6 TREAT tests U-Fs in 316SS (9rods) & U-Zr/U-Pu-Zr in D9/HT9 (6 rods)	18 RBCB tests; 30 breached rods 4 slow ramp tests 9 TREAT tests MOX in 316SS (14 rods) & HT9 (5 rods)	10 TREAT tests (10 rods; Na or He bond); ≤ 3-6 times TOP margins to breach Loss-of-Na bond test; RBCB for 100 EFPD; Centerline melting test

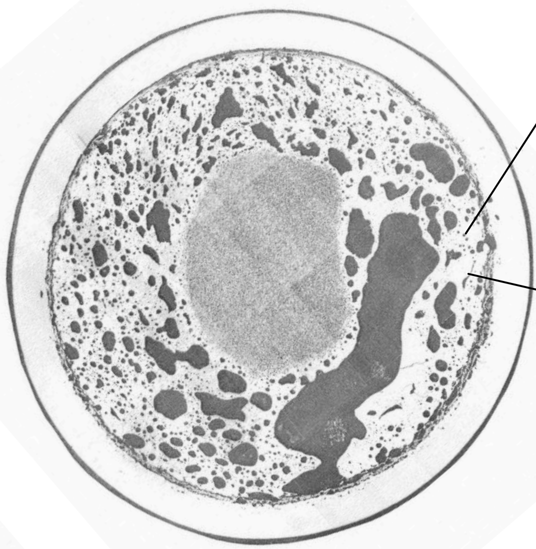
Transient Fuel Phenomena

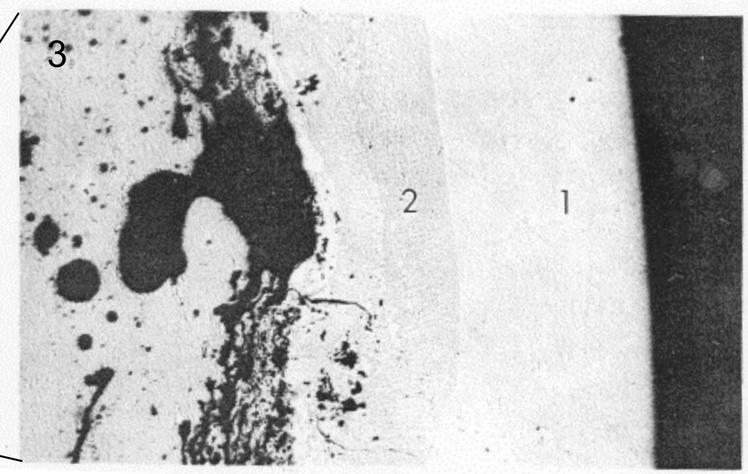
- **Metallic Fuels**
  - Pre-failure Behavior
    - *Substantial axial expansion*
    - *Cladding strain due to gas pressure*
    - *Possible fuel-cladding liquefaction*
  - Failure Behavior
    - *Failure generally near top of fuel column*
    - *Stress rupture due to gas pressure in cladding thinned by eutectic-like penetration and weakened at high temperature*
  - Post-failure Behavior
    - *Possible fuel injection into coolant*
    - *Low stored energy, no reaction with coolant, some local sodium voiding*
- **Oxide Fuels**
  - Pre-failure Behavior
    - *Axial relocation (apparently, upward axial motion)*
    - *Cladding strain due to FCMI and gas pressure*
  - Failure Behavior
    - *Failure generally in upper 1/3 of fuel column*
    - *Cladding melt-through with gas pressure and FCMI, cladding weakened at high temperature*
  - Post-failure Behavior
    - *Fuel dispersal into coolant*
    - *Relatively high stored energy, reaction with coolant, local sodium voiding*

Transient Phenomena—Metallic Fuels
Fuel/Cladding ‘Eutectic’ Formation

U-10Zr / HT9 at 800°C, 1 hr

(from H. Tsai, et al, 1990)





1 - Unaffected cladding
2 - Fuel/cladding solid-state interaction
3 - Fuel/cladding liquid phase

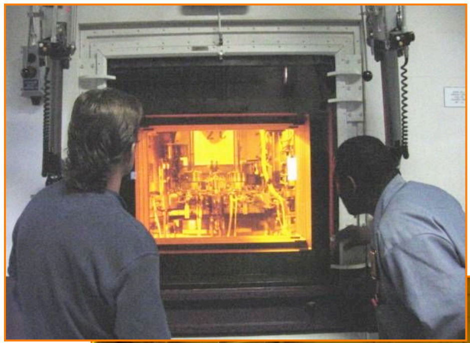
Metallic and MOX Fuels—Summary

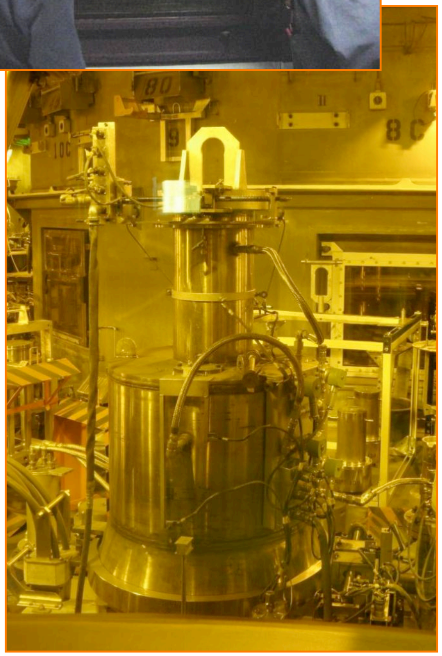
- **Metallic Fuels (U-Zr or U-Pu-Zr)**
  - Acceptable performance and reliability demonstrated up to 10 at.% burnup, with capability demonstrated to 20 at.% burnup
  - Robust overpower capability demonstrated in TREAT tests: ~ 4 to 5x's nominal power; failures near top of fuel column; pre-failure axial expansion
  - Performance issues typically creep rupture at high burnup, accelerated due to FCCI.
  - Performance phenomena with U-Fs, U-Zr & U-Pu-Zr are the same. Burnup, temperature and cladding performance are key variables
- **MOX Fuels**
  - Acceptable performance and reliability demonstrated up to 10 at.% burnup, with capability demonstrated to 20 at.% burnup
  - Robust overpower capability demonstrated in TREAT tests: ~ 3 to 4x's nominal power; well above primary and secondary FFTF trips; failures near core mid-plane; pre-failure axial fuel motion
  - Performance issues typically creep rupture at high burnup, accelerated due to FCMI (and FCCI if O/M not controlled).
- **Metallic and MOX fuel performance in SFRs are both well known, with good experience in the US (MOX fuel in France, Japan)**

Experience with Fuels Containing Minor Actinides

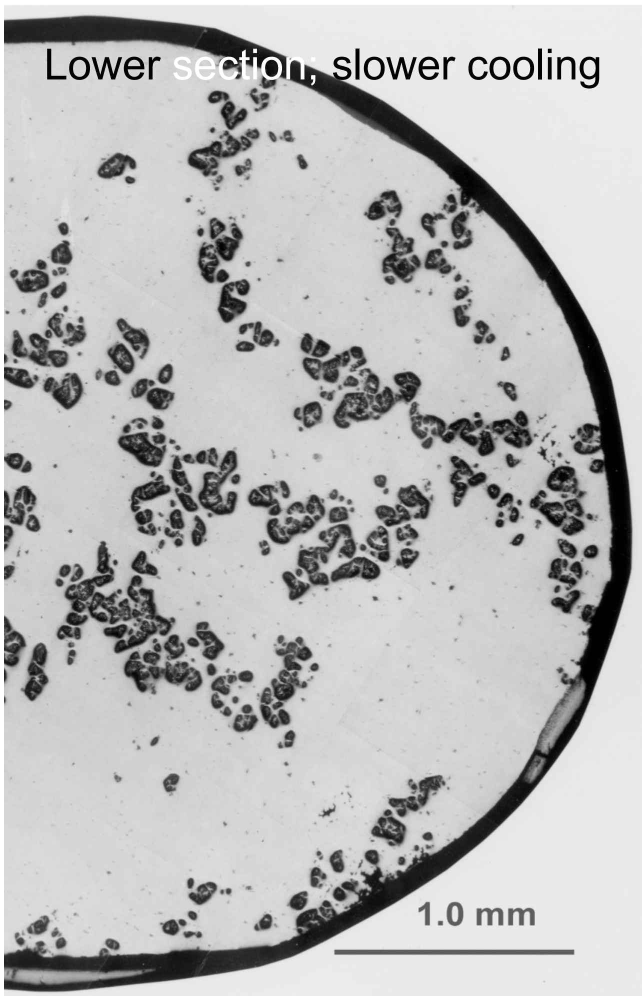
SFR Transmutation Fuels with Minor Actinides (MAs) and Rare Earth (RE) Fission Products

- **Unique Features of SFR Transmutation Fuels**
  - Pu content, which depending on CR selected my be higher than historic database (with corresponding decrease in U content)
  - Minor actinides (Am, Np, Cm) present in significant quantities
  - Rare earth fission product (La, Ce, Pr, Nd) carry-over from recycle step may be non-trivial
- **Gives Rise to Challenges and Unknowns**
  - Need for remote fuel fabrication
  - Likely need for new fabrication methods (e.g., due to Am volatility; waste minimization, etc.)
  - Effects on fuel performance must be determined

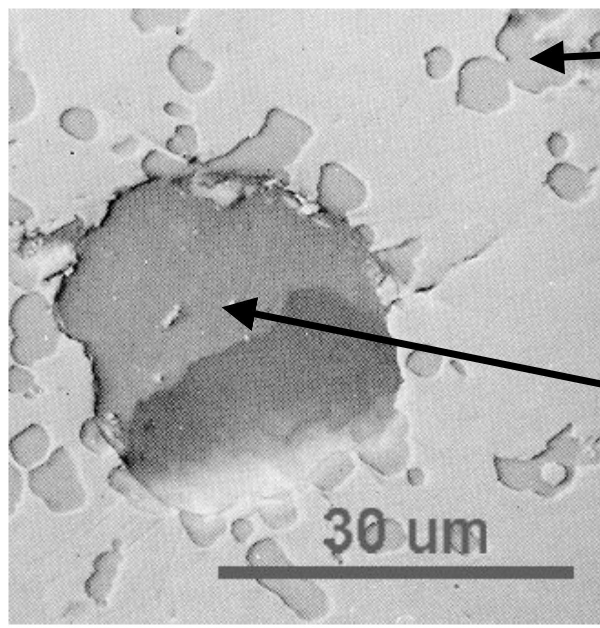




Metallic Fuel with MA—X501 Fabrication



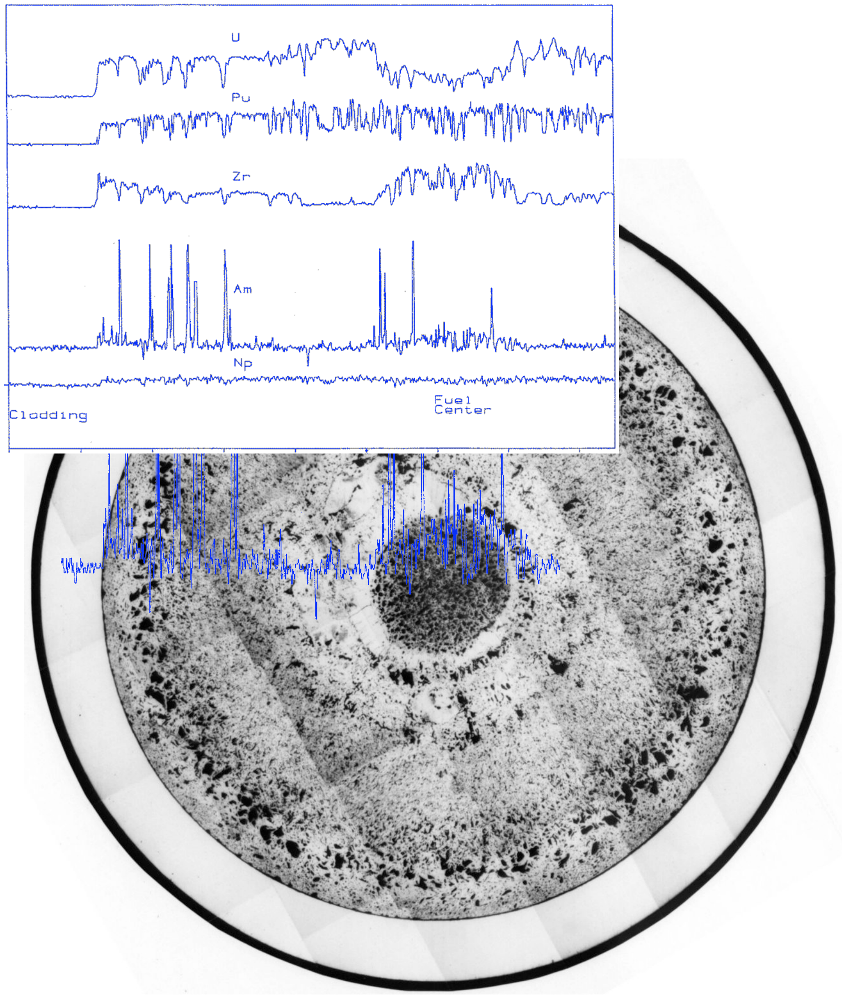
- U-20.2Pu-9.1Zr-1.2Am-1.2Np
- Injection cast at 1450°C
- Inhomogeneous microstructure
- Am and Np segregate to phases with variable composition

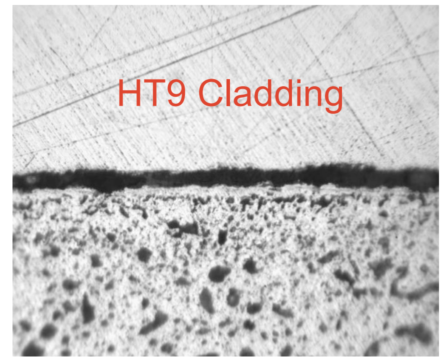


6 U
3 Pu
6 Np
86 Zr

21-47 U
14-49 Pu
9-19 Zr
0-25 Am
0-18 Np
Impurities

Metallic Fuel with MA—X501 Irradiation

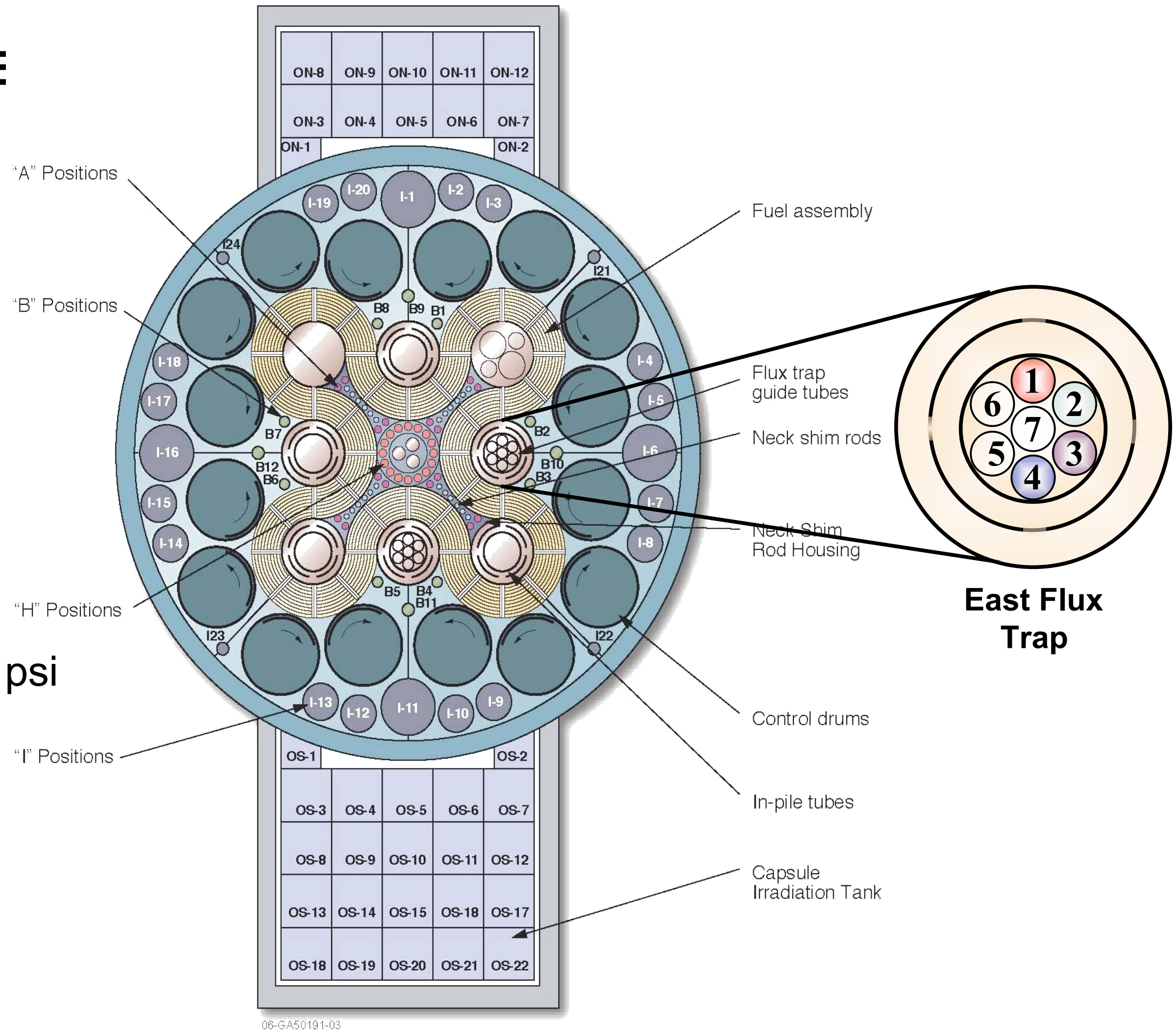




- LHGR = 450 W/cm
- PICT = 540°C
- Burnup = 7.6%
- ^{241}Am transmutation = 9.1%
- Gas Release
  - Fission gas = 79%
  - Helium = 90%

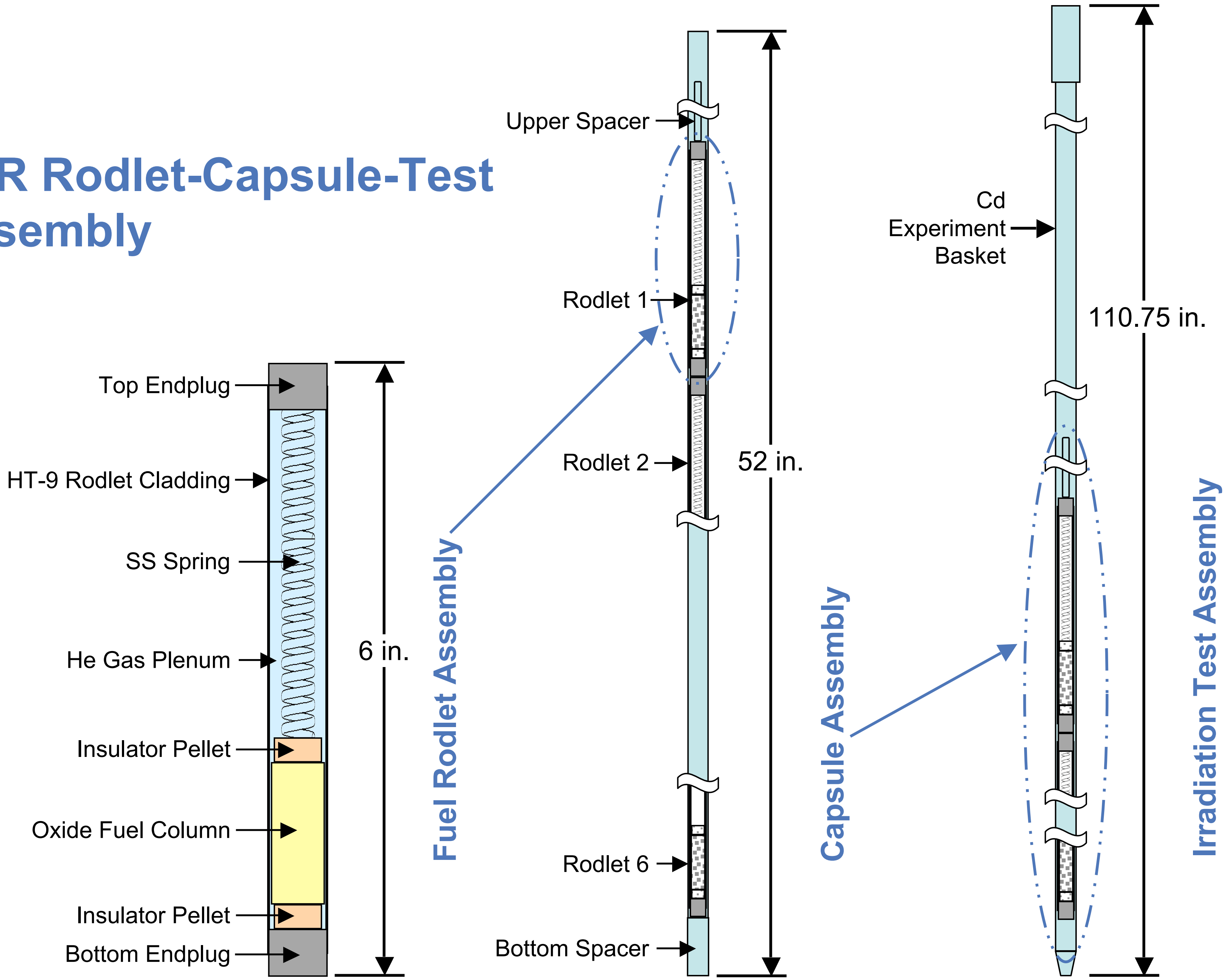
AFCI Fuels Testing in the ATR East Flux Trap

- **4 Capsule Positions in E**
  - Cd shrouds in 1,2,3,4
  - 6 rodlets per capsule
  - 24 rodlets irradiated simultaneously
- **Capsule Limits**
  - LHGR ≤ 500 W/cm
  - PICT ≤ 650°C
  - Capsule pressure ≤ 975 psi



East Flux Trap

ATR Rodlet-Capsule-Test Assembly

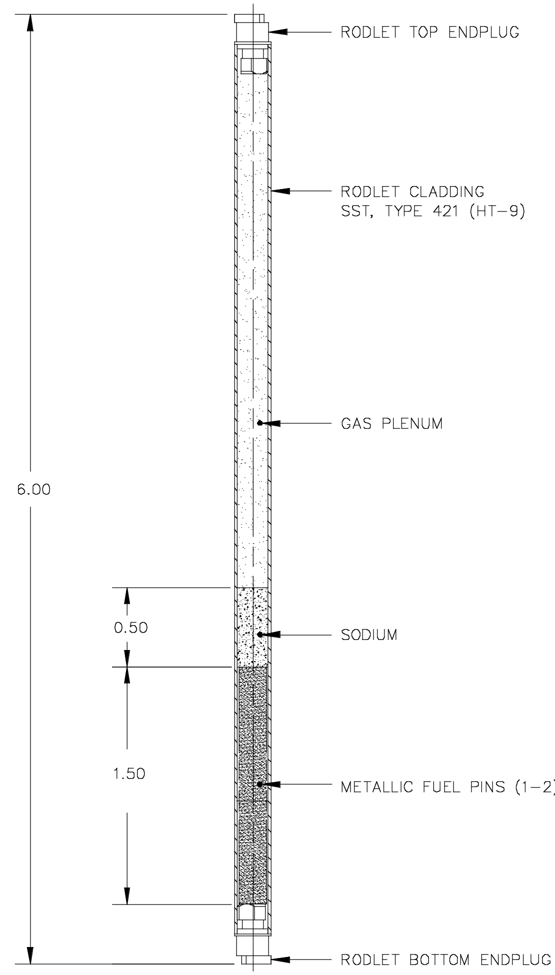


AFC-2A,B Currently Under Irradiation in the ATR

AFC-2A,B Test Matrix

Rodlet	AFC-2A&B
1	U-20Pu-3Am-2Np-15Zr
2	U-20Pu-3Am-2Np-1.0RE-15Zr
3	U-20Pu-3Am-2Np-1.5RE-15Zr
4	U-30Pu-5Am-3Np-1.5RE-20Zr
5	U-30Pu-5Am-3Np-1.0RE-20Zr
6	U-30Pu-5Am-3Np-20Zr

RE=6% La, 16% Pr, 25% Ce, 53% Nd



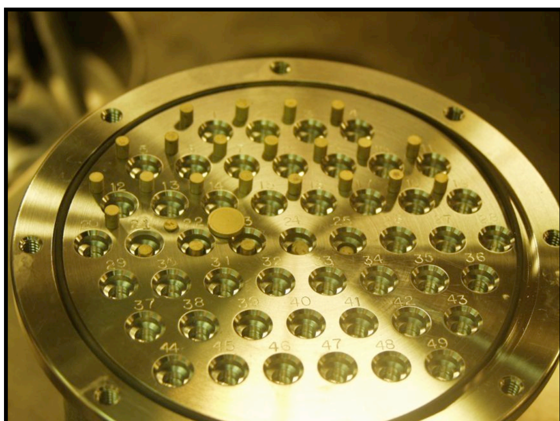
AFC-2A,B Test Objectives

- LHGR = 350 W/cm; PICT = 550°C
- Burnups of 10 at.% (2A) and 25 at.% (2B)
- Group recovery of 30 year-cooled PWR TRU
- Effect of RE fission product carry-over on FCCI

AFC-2C,D Currently Under Irradiation in the ATR

- **AFC-2C,D Test Matrix**

Rodlet	AFC-2C&D
1	$(U_{0.75},Pu_{0.20},Am_{0.03},Np_{0.02})O_{1.95}$
2	$(U_{0.80},Pu_{0.20})O_{1.98}$
3	$(U_{0.75},Pu_{0.20},Am_{0.03},Np_{0.02})O_{1.98}$
4	$(U_{0.75},Pu_{0.20},Am_{0.03},Np_{0.02})O_{1.95}$
5	$(U_{0.80},Pu_{0.20})O_{1.98}$
6	$(U_{0.75},Pu_{0.20},Am_{0.03},Np_{0.02})O_{1.98}$



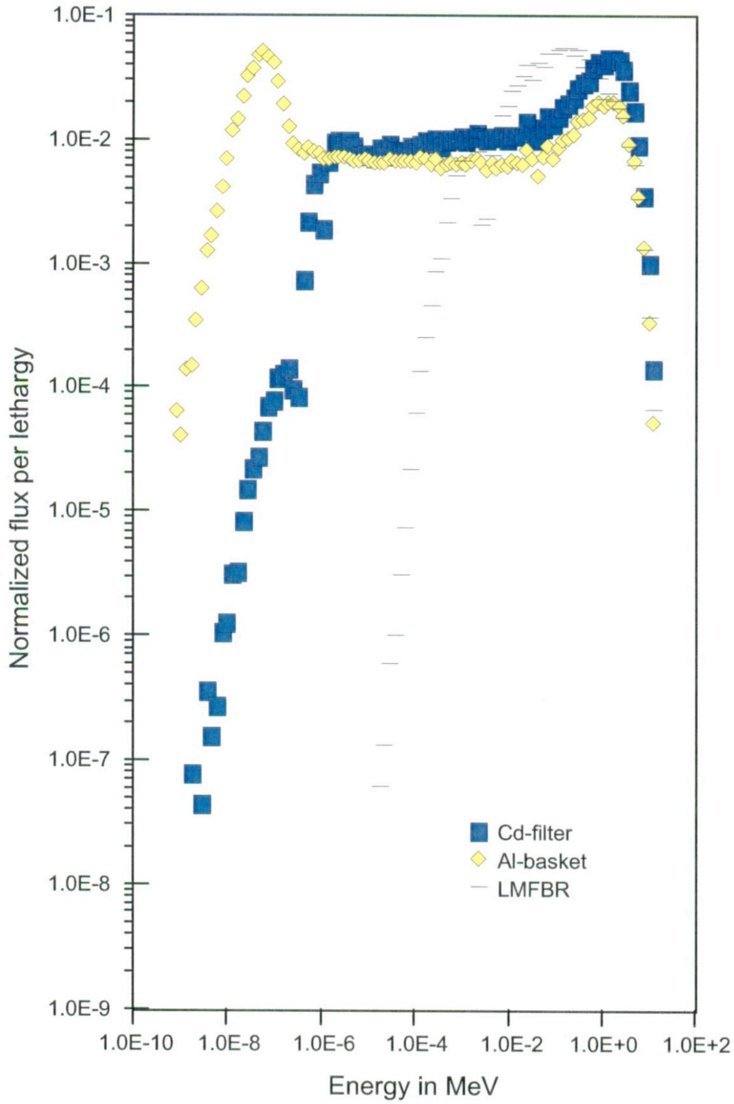
- **Test Conditions**
  - LHGR = 350 W/cm
  - PICT = 550°C
  - Group recovery of 30 year-cooled PWR TRU
- **Test Objectives**
  - Study effect of O/M on FCCI
  - Include MOX as control
  - High CR (20% Pu) for initial oxide test
  - Discharge criteria

    2C: ≥ 10 at.% burnup

    2D: ≥ 25 at.% burnup

Comparison of Spectra (ATR vs. LMFBR)

- **ATR Neutron Energy Spectrum**
  - Highly thermal spectrum in EFT with no neutron filter
  - Unaltered spectrum will result in significant self-shielding in dense, highly-enriched fuels
- **Cd-shroud Integral with Experiment Basket**
  - Efficient removal of neutrons with energies below cadmium cut-off
- **Resulting Spectrum**
  - Filtered spectrum in experiment does not have prototypic fast neutron component
  - Epi-thermal component responsible for most fissions; much more penetrating than thermal neutrons
  - Test fuels are free of gross self-shielding

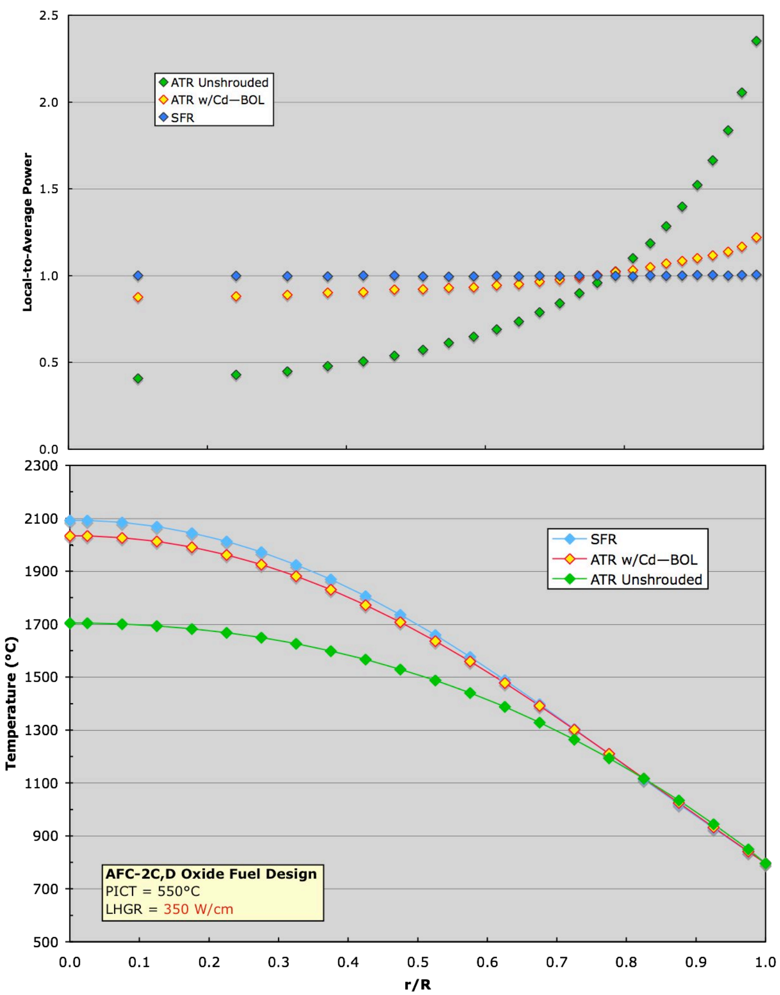


Radial Flux Depression and Temperature Profiles in Test Fuels

- **How prototypic are AFC rodlets irradiated in the ATR?**
  - Assessed by analysis
  - Radial power profiles calculated w/MCNP
  - Depletion in fuel and Cd shroud calculated w/ORIGEN (MCWO)
  - 1-D thermal analysis using radial powers

- **Resulting temperatures for AFC-2C,D oxide rodlets**
  - 3 cases: SFR, unshrouded ATR, ATR w/Cd shroud
  - w/Cd shroud, peak-to-avg power at fuel periphery is 1.22; fuel central temperature 58°C less than SFR (~400°C less for unshrouded case)

Radial Power & Temperature Profiles



Summary

- **SFR Fuels Experience in the US**
  - Fuel Types
  - Fuel Performance Issues
  - Experience/Testing
- **Experience with Fuels Containing Minor Actinides**